U. S. NUCLEAR REGULATORY COMMISSION

REGION V

Report No. 50-344/90-29

Docket No. 50-344

License No. NPF-1

Licensee: Portland General Electric Company
121 S.W. Salmon Street
Portland, OR 97204

Facility Name: Trojan

Inspection at: Rainier, Oregon

Inspection conducted: September 2 - October 6, 1990

Inspectors: R. C. Barr
Senior Resident Inspector, Trojan

J. F. Melfi
Resident Inspector

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Project Inspector

Approved By: P. J. Morrill 11/5/90
P. J./Morrill, Chief Date Signed
Reactor Projects Section 1

Summary:

Inspection on September 2 - October 6, 1990 (Report 50-344/90-29)

Areas Inspected: Routine inspection of operational safety verification, maintenance, surveillance, event follow-up, system engineering, and open item follow-up. Inspection procedures 30703, 40500, 56700, 61700, 61726, 62703, 71707. 90712, 92700, 92701, and 93702 were used as guidance during the conduct of the inspection.

Safety Issues Management System Items

II.F.2.2, "Instrumentation for Detection of Inadequate Core Cooling - Subcooling Meter," and II.F.2.4, "Instrumentation for Detection of Inadequate Core Cooling - Install Additional Instrumentation," are closed.

Results

General Conclusions and Specific Findings

This inspection identified two weaknesses in the licensee's Surveillance Program (Section 9). Surveillance procedures do not always establish appropriate acceptance criteria to verify component operability, and adequate administrative controls are not implemented to ensure surveillances are performed as a result of system or component change of status.

This inspection also identified two instances (Section 4 and Section 5) where deficiencies that were tracked by licensee open item tracking systems were inappropriately closed out. One of these items was a repeated level V violation and therefore resulted in a Notice of Violation.

This inspection identified operator knowledge deficiencies in the areas of generator breaker operation (Section 6) and inservice testing requirements (Section 5).

This inspection identified weaknesses in the licensee root cause evaluation program for non-safety related events (Section 6).

Significant Safety Matters

None.

Summary of Violations and Deviations

One cited (section 5) and one Non Cited Violation were identified (section 9).

Open Items Summary

Seven LERs (Section 7), three enforcement items and one Temporary Instruction item (Section 8) were closed. One open item was identified concerning the adequacy of the licensee's verification of shutdown margin (section 9).

DETAILS

1. Persons Contacted

   a. Portland General Electric

   *J. E. Cross, Vice President Nuclear
   *W. R. Robinson, Plant General Manager
   T. D. Walt, General Manager, Technical Functions
   G. D. Hicks, General Manager, Plant Support
   C. K. Seaman, General Manager, Nuclear Quality Assurance
   *C. P. Yundt, General Manager, Trojan Excellence
   *M. J. Singh, Manager, Plant Modifications
   J. D. Reid, Manager, Quality Support Services
   *J. W. Lentsch, Manager, Personnel Protection
   *A. R. Ankrum, Manager, Nuclear Security
   *J. A. Reinhart, Acting Manager, Operations
   *M. W. Hoffman, Manager, Nuclear Safety and Regulation
   W. F. Peabody, Manager, Nuclear Plant Engineering
   M. B. Lackey, Manager, Planning and Control
   J. F. Whelan, Manager, Maintenance
   S. A. Bauer, Branch Manager, Nuclear Regulation
   *J. Mody, Branch Manager, Plant Systems Engineering
   *D. L. Nordstrom, Branch Manager, Quality Operations
   *J. J. Taylor, Branch Manager, PM/EA
   G. L. Rich, Branch Manager, Radiation Protection
   W. O. Nicholson, Branch Manager, Operations
   R. L. Russell, Outage Manager
   *J. A. Benjamin, Supervisor, Quality Audits
   *W. J. Williams, Compliance Engineer

   b. Oregon Department of Energy

   A. Bless, Resident Engineer

   The inspectors also interviewed and talked with other licensee employees during the course of the inspection. These included shift supervisors, reactor and auxiliary operators, maintenance personnel, plant technicians and engineers, and quality assurance personnel.

   *Denotes those attending the exit interview.

2. Plant Status

   At the beginning of the inspection period, the facility was in Mode 1 at 100% power. At 4:59 pm, on September 24, 1990, due to a condenser tube leak, the reactor was shutdown (Section 6). The condenser tube was plugged and the reactor restarted at 3:48 am on September 26, 1990. At 1:50 pm, September 26, 1990, reactor power was reduced from 8% to 2% to evaluate another suspected condenser tube leak. At 1:11 am, on September 27, 1990, the reactor was returned to 8% power after repairing the tube leak. At 2:47 am, while attempting to parallel the main generator to the power distribution system, the main generator breaker malfunctioned resulting in the main generator being paralleled out of phase. At 8:29 am, the reactor was shutdown to Mode 3 to inspect the main generator, turbine and generator output breaker. At 1:06 am, on October 6, 1990, the reactor was restarted with 100% reactor power being reached at 8:59 pm. The inspection period ended with the facility at 100% power.

3. Operational Safety Verification (71707)

During this inspection period, the inspectors observed and examined activities to verify the operational safety of the licensee's facility. The observations and examinations of those activities were conducted on a daily, weekly or biweekly basis.

Daily the inspectors observed control room activities to verify the licensee's adherence to limiting conditions for operation as prescribed in the facility Technical Specifications. Logs, instrumentation, recorder traces, and other operational records were examined to obtain information on plant conditions, trends, and compliance with regulations. On occasions when a shift turnover was in progress, the turnover of information on plant status was observed to determine that pertinent information was relayed to the oncoming shift personnel.

Each week the inspectors toured the accessible areas of the facility to observe the following items:

(a) General plant and equipment conditions.
(b) Maintenance requests and repairs.
(c) Fire hazards and fire fighting equipment.
(d) Ignition sources and flammable material control.
(e) Conduct of activities in accordance with the licensee's administrative controls and approved procedures.
(f) Interiors of electrical and control panels
(g) Implementation of the licensee's physical security plan.
(h) Radiation protection controls.
(i) Plant housekeeping and cleanliness.
(j) Radioactive waste systems.
(k) Proper storage of compressed gas bottles.

Weekly, the inspectors examined the licensee's equipment clearance control with respect to removal of equipment from service to determine that the licensee complied with technical specification limiting conditions for operation. Active clearances were spot-checked to ensure that their issuance was consistent with plant status and maintenance evolutions. Logs of jumpers, bypasses, caution and test tags were examined by the inspectors.

Each week the inspectors conversed with operators in the control room, and with other plant personnel. The discussions centered on pertinent topics relating to general plant conditions, procedures, security, training and other topics related to in-progress work activities.

The inspectors examined the licensee's Corrective Action Program (CAP) to confirm that deficiencies were identified and tracked by the system.

Identified nonconformances were being tracked and followed to the completion of corrective action.

Routine inspections of the licensee's physical security program were performed in the areas of access control, organization and staffing, and detection and assessment systems. The inspectors observed the access control measures used at the entrance to the protected area, verified the integrity of portions of the protected area barrier and vital area barriers, and observed in several instances the implementation of compensatory measures upon breach of vital area barriers. Portions of the isolation zone were verified to be free of obstructions. Functioning of central and secondary alarm stations (including the use of CCTV monitors) was observed. On a sampling basis, the inspectors verified that the required minimum number of armed guards and individuals authorized to direct security activities were on site.

The inspectors conducted routine inspections of selected activities of the licensee's radiological protection program. A sampling of Radiation Work Permits (RWP) was reviewed for completeness and adequacy of information. During the course of inspection activities and periodic tours of plant areas, the inspectors verified proper use of personnel monitoring equipment, observed individuals leaving the radiation controlled area and signing out on appropriate RWP's, and observed the posting of radiation areas and contaminated areas. Posted radiation levels at locations within the fuel and auxiliary buildings were verified using both NRC and licensee portable survey meters. The involvement of health physics supervisors and engineers and their awareness of significant plant activities was assessed through conversations and review of RWP sign-in records.

The inspectors verified the operability of selected engineered safety features. This was done by direct visual verification of the correct position of valves, availability of power, cooling water supply, system integrity and general condition of equipment, as applicable.

No violations or deviations were identified.

4. Maintenance (62703)

Refueling Water Storage Tank (RWST) Heaters

The RWST, which supplies borated makeup water to the reactor during design basis accidents, is an uninsulated stainless steel tank located outdoors within the protected area. Technical Specification (TS) 3.5.5 establishes a minimum RWST temperature of 37 degrees F. (see FSAR Table 6.3-4) to ensure the boric acid additive does not precipitate. To maintain the temperature of RWST water above 37 degrees F., five 80 kw heaters are arranged symmetrically around the tank. The heaters are energized by a temperature element when the temperature falls below 40 degrees F. The inspector observed performance of Maintenance Request (MR) 90-8660, the preventative maintenance requirement that verifies RWST heater operability, and verifies the acceptability of the design capacity of the RWST heaters.

With respect to the observed maintenance, MR 90-9636 had technicians megger the heaters, measure the phase to phase and phase to ground resistances, measure the current flowing through the heaters when energized, and perform a visual inspection of the control center and junction box. The maintenance was performed by trained maintenance personnel using calibrated equipment. Radiological work practices were observed. The heaters were verified to function as designed. Because MR 90-8660 was designated as non-quality related and the RWST is safety related, the inspector examined the Trojan criteria for non-quality related work. The Trojan Nuclear Quality Assurance Manual (PGE 8010) defines quality related as "those activities, services, and equipment associated with safety-related structures, systems and components such as Environmental and effluent monitoring, Technical Specification monitoring ...". Because RWST heaters only maintain temperature and are not part of temperature monitoring, it appeared MR 90-9636 was correctly designated. As a result of the questions asked by the inspector, the licensee examined the RWST temperature indicator. The licensee determined the indicator was not properly characterized as quality related. A CAR was generated to correct the deficiency. The inspector also noted the fasteners to the junction box, in which the heaters were electrically connected, were corroded. This could, in the future, result in moisture intrusion. The licensee replaced the fasteners.

With respect to verifying RWST heater design capacity, the inspector identified that the Design Basis Document (DBD) did not include this calculation. The licensee obtained a copy of the calculation from the architect-engineer (A-E), however, the licensee could not verify the acceptability of the calculation because the calculation analysis technique was not included with the calculation. Consequently, the licensee performed an independent design calculation that verified the installed RWST heater capacity was adequate for the design assumptions. The licensee plans on revising the DBD to include the new calculation.

No violations or deviations were identified

Boric Acid Transfer Pump Motor Preventative Maintenance

The Boric Acid Transfer Pump (BATP) assists in reactivity control by pumping concentrated boric acid solution from the Boric Acid Storage Tanks (BASTs) to the Chemical and Volume Control System (CVCS). Technical Specifications that establish BATP requirements are 3.1.2.2.a, 3.1.2.5 and 3.1.2.6.

On September 11, 1990, the inspector observed licensee craftsmen perform preventative maintenance, in accordance with MR 90-9636, on the Boric Acid Transfer Pump. MR 90-9636 included motor meggering, greasing and inspecting. The inspector observed that craftsmen used calibrated equipment, followed procedures and observed quality control holdpoints. While performing the motor inspection, the craftsmen repaired damaged motor ventilation screens and generated a MR to repair cracked concrete under the motor baseplate of the east BATP.

In addition to the Resident Inspector observing the maintenance, a licensee Quality Control (QC) inspector also observed the performance of the work. Because the QC inspector was aware of previous lubricant issues, the inspector requested verification that the lubricant being used was an approved lubricant. The QC inspector found that the lubricant was acceptable but that an unapproved lubricant manual was used to select the lubricant. The QC inspector planned to document the finding with a CAR. Subsequently, she found a similar problem had been identified in Non-Conforming Activity Reports (NCARs) P88-015 and P89-014, revision 1, and Non-Conformance Report (NCR) 89-006. None of these reports had yet been closed, therefore, the CAR was not issued.

In discussions with licensee QA personnel, the NRC inspector found that several action dates on the NCARs and NCR had passed without responsible personnel recognizing that the action date had passed. For instance, the licensee had intended to perform an engineering evaluation of vendor recommendations by August 31, 1990, and to have a site lubrication manual implemented by September 21, 1990 (Memo CPY 059-90). The licensee switched to one vendor (Shell) for lubricants and is in the process of evaluating lubricant substitutions. The licensee acknowledges actions to resolve outstanding NCARs and CARs on lubricants have not been timely. The licensee has since established a tracking system to identify the number and responsible organization for overdue NCRs, NCARs and CARs. At the October 15, 1990 exit, the licensee stated that missed commitment dates for closure of actions on these items is no longer being tolerated. The inspectors will, through routine inspection, continue to monitor NCR and CAR closure.

No violations or deviations were identified.

5. Surveillance (61726)

The Component Cooling Water (CCW) System provides closed loop cooling for various Engineered Safety Feature components. Technical Specifications (TS) 4.7.3.1 and 4.0.5 require the pumps be tested per Section XI of the American Society of Mechanical Engineers (ASME) IST code. The IST code requires that the pump flowrate, vibration, and differential pressure (dP) be measured quarterly and bearing temperatures be measured annually. Periodic Operating Test (POT) 8-1, "Component Cooling Water System, Pump and Valve Inservice Testing," implements this testing.

The inspector observed the quarterly B CCW system pump test and the annual bearing temperature test of POT 8-1. The inspector noted the test steps were performed in order and the portable instrumentation used was within its calibration interval. The testing was done with qualified personnel. The operators complied with IWP 3500(b) by obtaining three readings approximately 10 minutes apart. The surveillance frequency was met. The surveillance summed flow from the three loop flows and, to obtain dP across the pump, the discharge and suction pressure was measured. The rated flow and pump dP were compared to the original pump curve. The licensee concluded the pump's performance was acceptable. The inspector observed that two of the installed Flow Indicators (FIs) were beyond their calibration interval (indicated by the calibration stickers) and the range scale of two instruments appeared to be outside IST requirements.

With respect to the two installed plant instruments, (FI-3207, last calibration 7/13/88 and FI-3208, last calibration 9/6/88), that exceeded the two year calibration interval, the inspector determined, by reviewing calibration records, that the instrumentation was still within the grace period allowed by the technical specifications (1.25 times the frequency interval, or 3.25 times three intervals). Additionally, these instruments did not have a history of drifting out of calibration. While conducting this test, the operators did verify that the portable instrumentation used was within its calibration interval, but did not verify that the installed instrumentation was within its calibration frequency. According to the personnel conducting the test, the Maintenance Department was responsible for ensuring field instrumentation was calibrated. The licensee concluded the existing controls for verification of instrument calibration were adequate.

With respect to the adequacy of range scales, the inspector noted the scale for FI-3207 was non-linear and read from 0 to 3000 gallons per minute (gpm) in 50 gpm divisions. The first scale reading was 500 gpm and the instrument was reading less than 500 gpm. ASME code IWP Part 4120 requires "the full scale range of each instrument shall be three times the reference value or less." The reference value was a flow of less than 500 gpm. Consequently, the gauge appeared not to meet the requirements of the code. The safety significance of this noncompliance is minimal because the flow values measured by this instrument were a small percentage of the total flow. The total CCW flow value was acceptable.

Through review of licensee records, the inspector identified that this issue had been previously raised, but not effectively resolved. Request For Evaluation (RFE) 4181 had been submitted on November 30, 1987, by the Operations department to evaluate this gauge against Code Requirements. The RFE was evaluated on January 5, 1990, and stated that this issue was covered by RFE 6152 (issued 1/10/89). RFE 6152 was evaluated on January 15, 1990, and the disposition of the RFE was to use a temporary gauge with the appropriate scale at the existing location. However, due to an apparent weakness in the RFE process, the installed gauge instead of a temporary gauge was used. The licensee is evaluating the cause of this error.

In discussions with licensee IST engineers, the inspector learned that the issue of appropriate ranges for several instruments had been identified in NRC inspection report 50-344/85-20. It was also recognized in the report that the use of one flow meter for CCW flow would simplify the surveillance. At the conclusion of this inspection period, the licensee was in the process of installing one flow meter to measure CCW flow so that three indicators would not have to be summed to obtain total flow. As a result of inspection 50-344/85-20, the licensee initiated Request for Design Change (RDC) 85-057 to install various indicators with the proper range. The RDC had various Design Change Packages (DCPs) associated with it and the inspector was informed that the DCP (Number 3) associated with the CCW flow had not been properly tracked and had not been implemented. The NRC open item associated with this issue had been closed in 1987 on the basis that DCP 1 would be implemented during the 1987 refueling outage. The majority of the work on DCP 1 to change instruments was completed and an engineer noted that the work was substantially complete. Due to an incomplete review, the RDC was subsequently signed off as complete. In March 1990, the licensee identified that the total CCW flow instrument, which was to be installed by RDC 85-057, had not been installed and attempted to get the DCP implemented. However, the actions necessary to complete RDC 85-057 before the surveillance was required were not successful.

The inspector identified a similar concern with the range and accuracy of the temperature instrument, a pyrometer, used to measure B CCW pump bearing temperature. The pyrometer had a 0 to 500 degree F. scale in 10 degree increments. The reference temperature was approximately 70 degrees F. Part IWP 4120 of the ASME code states that "the full-scale range of each instrument shall be three times the reference value or less." The historical value for the "reference value" for this temperature is approximately between 50 and 70 degrees F. The full-scale instrument range to use in this case would have been between 150 and 210 degrees F. The inspector also verified that no program existed to train operators on gauge accuracy requirements. The improper full-scale range is an apparent violation (50-344/90-29-01).

The licensee stated that they met the intent of the ASME code since the instrument used was more accurate than the code required. The licensee based this conclusion on the required accuracy for temperature instruments which is $\pm$ 5% (IWP 4110), and if the full scale required value is 150 degrees F., this would imply an accuracy of $\pm$7.5 degrees F. The instrument used was calibrated to a greater accuracy than $\pm$7.5 degrees F. The licensee stated that they had issued an exemption request to the NRC to delete the bearing temperature measurements, as was allowed by a code change, but the NRC had not yet acted on the request. Also, the licensee's procedures do not include guidance on required instrument ranges for bearing temperature measurements. The licensee intended to have a dedicated temperature instrument with a range of 150 degrees F. for this surveillance put in the control room by 10/17/90. To address this violation of code requirements, the licensee also invalidated the test on bearing temperature and will reperform the test during the next quarterly surveillance.

One violation was identified.

6. Event Follow-up (93702, 62703, 40500)

Main Condenser Tube Leaks

On September 24, 1990, the reactor was shutdown to repair a suspected condenser tube leak in the C Main Condenser (MC). Licensee craftsmen found that the hard neoprene plug, which had been installed in July 1990 to plug the leaking tube, had failed. PGE root cause identification concluded the plug, which appeared to have had a cut on the inner plug surface propagating to the outer surface, failed as a result of a manufacturing defect or an installation introduced defect. To prevent future failures, the licensee identified the plug lot number, replaced all plugs of this lot, and changed the method of plug installation. The leak was repaired and the reactor restarted on September 26, 1990.

During the September 26 reactor startup, while at approximately 8% reactor power, condensate conductivity increased indicating another condenser tube was leaking. Management decided to reduce reactor power to 2%, maintain vacuum, drain the C condenser and evaluate and repair the leak. Licensee craftsmen and engineers found that the neoprene plug that had been installed on September 24 had again failed. Licensee engineers attempted to identify the location of the defect in the condenser tube by using a boroscope. However, due to condenser vacuum being maintained, there was too much turbulence to locate the defect. The turbulence was so great that the boroscope cabling was damaged in the process. The craftsmen again plugged the tube, however, a phenolic plug, with which the licensee had experienced previous problems, was used.

On September 27, 1990, after repairing the tube leak, reactor power was increased. While at approximately 8% power and attempting to parallel the main generator to the electric distribution system, the main generator output breaker malfunctioned resulting in the generator being paralleled out of phase. The reactor was shutdown to inspect the main generator, main generator output breakers, and the low pressure turbines.

During the shutdown, the main condenser was again drained to evaluate the effectiveness of the phenolic condenser plug. Licensee craftsmen and engineers found the plug loose. When the condenser was accessed from the shell side, the engineers found that the failed tube was completely severed approximately three inches from the inside face of the tube sheet. The engineers determined that the cause of the neoprene plug failures was the jagged edge of the failed tube which cut the plugs after installation. The severed tube was removed at the first support plate and the tube plugged with a stainless steel plug.

The inspectors reviewed the maintenance requests for the plugging activities described above, looked at the failed neoprene plugs, and discussed the plugging activities with licensee management, engineers and craftsmen. The inspectors concluded the licensee's root cause evaluation of the plug failures was weak. Following the plug failure of September 24, the licensee determined the failure was due to a plug defect or installation induced failure. The plug had a defect that appeared to be a cut, however, the source of the cut was not determined. The root cause evaluation did not identify that detailed plug installation procedures were not available. After the second neoprene plug failed, an attempt to examine the failed tube was aborted because the conditions required to examine the tube were not established. After the failure of the second neoprene plug, a phenolic plug, which had previously been abandoned, was used. Subsequent licensee inspection found that plug was loose because the tube had severed. These plug failures indicate the need for the licensee to improve their root cause program for non-safety related events.

Main Generator Output Breaker Malfunction

For reliability, the Trojan main generator has two 230 KV output breakers (V838 and V842) that are paralleled and connect the main generator to the regional power distribution system. The output breakers, type 45A, were manufactured by ITE. The breakers have been maintained by PGE's Substation Maintenance and Construction Division.

On September 27, 1990, at approximately 2:47 am, when attempting to parallel (synchronize) the main generator output, using V838, tc the electrical distribution system, the control operator, per procedure, took the breaker control switch to close. The breaker did not close within the prescribed band (five degrees on either side of vertical) on the synchroscope meter. Because the breaker did not shut, the operator repositioned the V838 control switch to trip, expecting the breaker to remain open. The breaker shut when the synchroscope was at approximately 120 degrees from the desired position (ie: 4:00 on a clock face), resulting in it tripping on overload. After consulting with the turbine-generator vendor, the licensee, at 8:15 am, September 27, 1990, shut down the reactor to inspect the turbine, the generator and the generator output breakers.

The licensee documented the event and the evaluation in Corrective Action Request (CAR) 90-5325. Licensee inspection of the main generator identified no damage. The inspection of the low pressure turbine identified one loose bucket on the outer most diaphram of the B low pressure turbine. The blade was tightened. The licensee concluded the loose blade was not caused by this event, but was a result of the blade attachment pins wearing. Licensee investigation of the generator output breaker found that the pilot valve that directs air to the pneumatic operator was fouled with a fine graphite like powder which most probably resulted in slow breaker operation. Additionally, one of the pilot valve's gaskets had a flat spot. The licensee replaced the pilot valve on both breakers. After the repairs, the breakers were time cycled and shut within the prescribed time interval (0.27 sec.). With respect to breaker operation, the licensee's evaluation found that the breaker control circuit functioned as designed. The breaker was designed not to reopen, even if a trip signal was demanded, until after the breaker closed. To correct operator knowledge deficiencies on operation of the generator output breakers, the licensee conducted training sessions for each shift of operators. Additionally, the licensee is reevaluating the preventative maintenance program for the V838 and V842 breakers for adequacy.

The inspectors reviewed CAR 90-5350 and observed selected portions of the licensee inspection of the turbine, generator and generator output breaker. In discussions with licensed operators, the inspectors learned that for a year or more, the generator output breakers have been closing slowly. However, the operators did not research the required closure time nor did they identify the concern formally through the various Trojan corrective action systems. The operators had expressed their concerns to the first and second levels of supervision but, they also did not explore the concern. Licensee corrective actions to repair the generator output breaker appeared appropriate.

No violations or deviations were identified.

7. Follow-up of Licensee Event Reports (LERs) [90712, 92700]

LER 90-23, Revision 0, (Closed), "Inadequate Temporary Procedure Revision Leads to Failure to Document Verification of Reactor Coolant Flow During Boron Dilution." The reactor coolant system boron concentration was reduced while system fill was in progress without verifying the rate of reactor coolant flow. The licensee concluded that an inadequate and improper temporary procedure revision allowed the boron dilution without ensuring that applicable Technical Specification Surveillance requirements were met. Licensee corrective actions included withdrawal of the temporary procedure revision that initiated this event, review and revision of other procedures used to adjust boron concentration, and installation of a permanent caution tag on the makeup control switch. Additionally, the licensee revised the process for making procedure revisions and expected to incorporate a data sheet for verification of reactor coolant flow into a daily operating procedure.

The inspector verified that the corrective actions specified by the licensee were complete. The inspector noted that the licensee determined that the data sheet for reactor coolant flow verification would be more appropriate in Periodic Operating Test (POT) 24-1, "Shift Operating Routines," than in POT 24-2, "Daily Operating Routines." The inspector reviewed the revision to POT 24-1 which added the flow verification data sheet. This item is closed based on the licensee's completed actions.

LER 90-24, Revision 0, (Closed), "Improper Technical Specification Interpretation Leads to Inadequate Procedure Revision and Incorrectly Performed Surveillances." The licensee determined that the procedure used to implement Technical Specification Surveillance 4.6.1.5, "Containment Systems Air Temperature," contained a discrepancy in tha the procedure did not ensure that the appropriate locations for monitoring temperature were selected. A revision to Periodic Operating Test (POT) 24-2 would allow the operators to select temperature monitoring points that were not intended by the surveillance requirements. The licensee determined the cause of the improper revision was a failure to literally interpret the applicable requirements. The corrective action was to revise POT 24-2 to correctly implement the Technical Specification requirements.

The inspector reviewed the revision to POT 24-2 that appropriately incorporated the requirements of Technical Specification Surveillance 4.6.1.5. The inspector also reviewed the revision with various operators and determined that the operators would perform the procedure correctly. This item is closed based on the licensee's completed action.

LER 90-25, Revision 0, (Closed), "Incorrect Interpretation of Regulatory Guide Results In Not Performing a Required Surveillance on Reactor Coolant Pump Flywheel Inspection." The licensee determined that Technical Specification requirements to examine reactor coolant pump flywheels in accordance with Regulatory Guide 1.14, "Reactor Coolant Pump Flywheel Integrity," were not being correctly implemented. The Regulatory Guide indicates that exposed surfaces of the flywheels are to have an examination at approximately 10 year intervals. The licensee had not performed the surface examination of the flywheels during the first 10 year interval due to an incorrect interpretation that the surfaces were not exposed because the flywheels were painted. Corrective actions included performance of an examination of accessible surfaces for all four reactor coolant pump flywheels and plans to revise PGE-1049, "Inservice Inspection Program Second Ten-Year Interval," to include the surface examination.

The inspector verified that the surface examination of the reactor coolant pump flywheels was performed during the last outage and confirmed that a commitment had been established to revise the inservice inspection program by December 14, 1990. This item is closed based on the licensee's completed and proposed corrective actions.

LER 90-26, Revision 0, (Closed), "Communication And Procedural Errors By Personnel Result In Actuation of An Engineered Safety Feature Component And Opening of The Reactor Trip Breaker." While performing time response testing of the Engineered Safety Feature Actuation System, a containment spray pump inadvertently started due to a temporary electrical jumper that had not been removed from a portion of the pump control circuit. The jumper had been installed for a previous portion of the time response test and was not required for subsequent testing. The procedural step that specified removal of the jumper had not been performed. The licensee concluded that the cause of the inadvertent containment spray pump start was inadequate continuity of work control as the testing spanned more than one shift.

Subsequent to the event described above, an unplanned actuation of the Reactor Protection System occurred during performance of reactor trip breaker time response testing. The unplanned opening of the reactor trip breaker occurred as a result of not following the procedure in a sequential manner. The reactor was in Mode 5 at the time of the testing.

The licensee suspended nonessential work as a result of the events. Corrective actions included (1) issuance of a revision to Administrative Order (AO) 4-2, "Use of Procedures," to require that procedural steps be performed in sequence unless the procedure indicates otherwise and (2) holding of meetings with plant personnel to communicate management's expectations on procedural compliance. Quality Assurance also performed surveillances to observe procedural compliance.

The inspector verified that the corrective actions were completed. As part of their routing followup inspection, the inspectors will continue to evaluate the licensee's implementation of procedural compliance requirements. This item is closed based on the licensee's completed actions.

LER 90-27, Revision 0, (Open), "Inadequate Implementation of a Programmatic Change In How a Technical Specification Surveillance Was To Be Met Results In a Missed Surveillance Due To An Inadequate Procedure." The licensee determined that a procedure, used to implement a Technical Specification (TS) requirement for a monthly surveillance of containment boundaries, did not include all of the boundaries. Periodic Operating Test (POT) 3-3, "Containment Penetration Valve Inservice Test," which implements TS 4.6.1.1.a.1, was found not to include a check of the positions for two steam generator blowdown system drain valves. After further investigation, the licensee determined that the POT did not list an additional ten valves that should be checked to satisfy the TS requirements. The licensee concluded that a change in the method of implementing the TS in 1988 along with an ongoing design modification resulted in omission of the valves from the POT. It is noted that operating procedures positioned the valves in the desired (shut) position.

Corrective actions included revising POT 3-3 to include the twelve valves that were not listed, plans to review drawings to ensure that vent, test and drain valves within a containment penetration boundary are listed in POT 3-3 or controlled as locked, and a containment design basis document review for physical verification of vent, test and drain valves within a containment penetration boundary.

The inspector verified that commitments were made to perform the drawing review and the design basis document review of containment penetrations. The inspector reviewed temporary change notice (TCN) 90-068 to POT 3-3. The TCN added eight valves to the list of containment boundaries; not 12 valves as reported by the LER. This discrepancy was brought to the attention of Plant Management at the exit meeting. This LER remains open pending licensee evaluation of the discrepancy between the reported and actual corrective action.

<u>LER 90-28, Revision 0, (Closed), "Auxiliary Feedwater Pump Control Instrumentation Was Not Seismically Mounted As A Result of Inadequate Work Instructions."</u> During a walkdown of the main control panels, the auxiliary feedwater differential pressure controllers were identified as being loose and not having mounting brackets installed. The manufacturer's seismic test of the controllers was performed with mounting brackets installed. The licensee declared both trains of auxiliary feedwater inoperable and performed a plant cooldown from Mode 3 to Mode 4 in order to comply with Technical Specification requirements. The licensee concluded that the most probable cause of the mounting brackets not being installed was inadequate work instructions. Corrective actions included installation of mounting brackets for the loose controllers, walkdown of the control panels for other potentially loose controllers, and commitments to train work planners on the need to include information on seismic qualification of equipment in work instructions.

The inspector confirmed that the licensee inspected the control panels for other loose controllers that are safety related. No additional loose controllers were found. The inspector also confirmed that training for the work planners was scheduled. The mounting bracket that was installed for the loose controllers was observed. The inspector noted that the controllers were secured to the mounting bracket with flexible cable-ties. While the use of the cable-ties provides a method of securing the controllers to the mounting bracket, the cable-ties could be cut and not replaced as the installation appears temporary. The licensee is designing a metal fixture for securing the controllers to the mounting bracket and has scheduled installation of the metal fixture during the

1991 Refueling Outage. This item is closed based on the licensee's completed and proposed corrective actions.

LER 90-29, Revision 0, (Closed), "Failure To Follow Procedures Results In Mode 3 Operation With Both Safety Injection Pumps' Automatic Start Capability Disabled." Both safety injection pump control switches were found in the pull-to-lock position when the plant was in Mode 3. With the switches in pull-to-lock, the pumps were not available for automatic starting, as required by the Technical Specifications for Mode 3 operations. After discovery, the switches were placed in the auto position. The licensee determined the switches had been mispositioned for approximately four and one-half hours as a result of an operator not following the procedure for transitioning from Mode 4 to Mode 3. Corrective actions taken by the licensee included counseling and discipline of the operator and reminding the Shift Supervisors and Assistant Shift Supervisors of their responsibilities during significant plant evolutions. Additionally, procedures for changing operational Modes were revised to include a check and sign off requirement for the Shift Supervisor to signify that all preparations have been made and all procedure steps are completed prior to changing Modes.

The inspector verified that the corrective actions were performed. The inspectors, as part of their routine inspection, will continue to evaluate the licensee's compliance with procedures. This item is closed based on the licensee's completed actions.

LER 90-31, Revision 0, (Closed), "Inadequate Test Procedure Results In Failure To Document Status Of Component Cooling Water Valves For Technical Specification Surveillance." The licensee determined that a Technical Specification requirement to cycle certain component cooling water system valves had not been adequately documented in procedures. The positions of two valves which isolate component cooling water to the letdown and seal water heat exchangers were not recorded prior to initiation of a containment isolation test signal. Thus, requirements to verify that the valves repositioned during the test were not documented. The licensee concluded that the cause of this event was personnel error in developing the procedures used to implement the Technical Specification requirements. The licensee declared the applicable train of the component cooling water system inoperable. The procedure was revised to properly document the testing of the valves and the test was reperformed. Subsequent to successful testing, the component cooling water system was declared operable. Corrective actions by the licensee also included plans to review surveillance procedures related to Engineered Safety Features Actuation System output relays to determine if any similar deficiencies exist. Long term corrective action is to be identified in future correspondence with the NRC.

The inspector verified that the licensee corrective actions were performed or have been scheduled. While changing the procedure to document testing of the component cooling water system valves, the licensee noted that their testing technique differed from the test as described in the Final Safety Analysis Report (FSAR). Specifically, the licensee's test requires installation of electrical jumpers around blocking relays in Solid State protection System (SSPS) cabinets C-40A and C-40B. Jumpering of the blocking relays allows component actuation from the SSPS cabinets. Section 7.3.2 of the FSAR does not address use of jumpers around blocking relays to actuate components from the SSPS cabinets.

The Corrective Action Report (CAR) for this event indicated that a long term corrective action is to evaluate the test technique vs. the FSAR. The inspector noted that the CAR did not have an attached action request for performance of the evaluation. The inspector also noted that the CAR had been through the first level of Quality Assurance review (Performance Monitoring and Event Assessment). The lack of an action request was brought to the attention of Plant Management during the exit meeting. This LER is closed based on the licensee's completed and scheduled corrective actions. The inspectors, during their routine inspection, will follow the licensee's actions to evaluate their test technique.

No violations or deviations were identified.

8. Followup of Notices of Violations and Temporary Instructions (92701)

Enforcement Item 50-344/90-02-01, (Closed), "Incomplete Final Safety Analysis Report (FSAR) Updates." The licensee, in the April 4, 1990, reply to the Notice of Violation, concluded that the cause of the violation was not providing Licensing Document Change Requests (LDCRs) for Design Change Packages (DCP) completed prior to May 2, 1986. PGE noted they had also recognized this concern during a December 29, 1989, Nuclear Quality Assurance Department (NQAD) audit, that also identified that Nuclear Department Procedure (NDP) 200-1, "Design Change Control," did not have a time requirement to submit the LDCR following the completion of a DCP.

As corrective actions, the licensee committed to revise the FSAR by August 31, 1990, for RDC 83-051, and by April 30, 1990, for RDC 76-068. On August 30, 1990, the licensee revised the April 4, 1990, response to relate that RDC 83-051, DCP-3 would be included in Amendment 14 vice by August 31, 1990. The licensee also committed to review all DCPs issued since January 1, 1988, to verify that the FSAR had been properly updated as a result of changes to the facility design, and based upon the results of the review, consider expanding the review scope to include earlier design changes. The licensee completed the review and determined the scope should be expanded to include all RDCs to verify the FSAR was updated as required.

The resident inspectors verified that RDC 76-068 was incorporated in FSAR Amendment 13; that NDP 100-5, "Preparation of Safety Evaluations Required by 10 CFR 50 and Trojan Technical Specifications," was revised; and that Nuclear Plant Engineers completed training on NDP 200-1. The resident inspectors, via routine inspection, will continue to review the FSAR revision process. Another example of the FSAR not being maintained current is discussed in Section 9 of this report. Based on the corrective actions taken, this item is closed.

Enforcement Item 50-344/90-11-02, (Closed), "Procedural Noncompliances While Conducting Maintenance on Motor Operated Valves." This enforcement action documented a three part violation of requirements not met during the licensee's conduct of maintenance on motor operated valves. In the July 6, 1990, response to the Notice of Violation, the licensee concluded the causes of the violation were a combination of personnel error and procedural inadequacy. The licensee also noted that ineffective communication between work groups and poorly organized work instructions contributed to the violation. As corrective actions, the licensee counselled personnel that did not follow procedures, revised procedures, issued a "Lessons Learned Summary" to all Nuclear Division employees, and conducted work group training on the lessons learned from this event. Additionally, PGE committed to monitor the CAR/Excellence Response Programs for the next six months to ensure corrective action requests are properly addressed (Commitment Tracking List Item 40624).

The inspectors verified that the training committed to was conducted, the procedural changes committed to were performed and the valves on which the maintenance was conducted were operable. Through routine inspection, the inspectors will continue evaluating the licensee's resolution of CARs. Based on licensee corrective actions, this item is closed.

Enforcement Item 50-344/90-21-01, (Closed), "Inattentive Fire Watch." This enforcement action documented a continuous fire watch not properly standing watch. In their September 25, 1990, response to the Notice of Violation, the licensee concluded the cause of the violation was personnel error. As corrective actions, the licensee terminated the fire watch's employment per company policy, reemphasized the need for attentiveness and the consequences of being found inattentive, established a mandatory policy for standing while on watch or wearing an anti-sleep device if sitting, changed the subject fire watch post to a one hour post vice a two hour post, and distributed an all employee memorandum that clarified the consequences of sleeping within the protected area.

The inspectors, through routine inspections, have noted fire watches complying with the mandatory policy of wearing anti-sleep devices while sitting. The inspectors, two months following the distribution of the all employee memorandum that clarified PGE's policy on sleeping within the protected area, polled seventeen employees to determine if the policy was understood. Based on only four of the seventeen employees correctly describing the policy, the inspector concluded the policy was not fully understood. The inspector shared the results of the polling with senior PGE management. Subsequently, the issue of inattentiveness has been discussed with all PGE Nuclear Division employees during work group meetings. The Vice President, Nuclear met with all Nuclear Division managers and stressed the importance of communicating significant and sensitive issues to their subordinates. Based on licensee corrective actions, this item is closed.

TI 2515/105, (Open), "Inspection of Licensee Activities in Peference to Bulletin 88-04, 'Potential Safety-Related Pump Loss.'" This Temporary Instruction (TI) was issued to verify the satisfactory implementation of NRC Bulletin (NRCB) 88-04. The Bulletin (issued May 5, 1988) requested that licensees investigate and correct two possible miniflow design concerns. These concerns were the possibility of dead-heading one or more pumps in systems with a common miniflow recirculation line and whether the capacity of the installed miniflow recirculation line was adequate for even one pump in operation. These concerns were also referenced in Information Notice (IN) 87-59, "Potential RHR Pump Loss," dated November 17, 1987

The licensee received IN 87-59 on November 11, 1987, and entered it to be evaluated in the Operating Experience Review (OER) Program. The evaluation was performed on April 24, 1989, with corrective actions to be completed as part of Action Plan 89-005, "Safety Related Pump Minimum Flow."

The licensee received NRCB 88-04 on May 9, 1988, and was requested to respond within 60 days of its receipt. The licensee responded on July 18, 1990, stating that only the Residual Heat Removal (RHR) pumps had the potential to interact in the manner described in the bulletin. The licensee also determined that the probability of strong/weak pump interaction was extremely low since the pumps were operating almost identically and any pump degradation would be identified during quarterly pump testing. The licensee also committed to respond on the long term actions taken on future dates and to provide the pump vendor information.

On September 29, 1989, PGE extended the Bulletin closure date to January 15, 1990, due to difficulty in obtaining pump data from the vendor. In a February 27, 1990 memorandum that described long term actions associated with the Bulletin, the licensee stated further delays had been incurred in obtaining the pump vendor information and tests would be conducted by July 30, 1990, to evaluate pump interaction with the results provided to NRC by September 30, 1990. The inspectors will continue following the PGE response to this bulletin.

TI 2515/65, "TMI Action Plan Followup."

TMI Item II.F.2, (Closed), "Instrumentation for Detection of Inadequate Core Cooling (ICC)." This Three Mile Island (TMI) action plan item required that the licensee provide on-line indication of Reactor Coolant saturation conditions. The two sub-items under this TMI item still open were to install a subcooling meter (II.F.2.2) and install additional instrumentation (II.F.2.4). Additional instrumentation was added to provide unambiguous indication of inadequate core cooling (ICC) which included the Reactor Vessel Level Instrumentation System (RVLIS), Sub-Cooled Margin Monitors (SMMs) and dedicated Core Exit Thermocouples (CETs). NUREG 0737, "TMI Action Plan," provides guidance on what to inspect to verify this item. Some of these requirements were also included in later revisions of Regulatory Guide (RG) 1.97, "Post-Accident Instrumentation." The following reviews and verifications were completed to close this item:

1) A review of Temporary Plant Test (TPT)-58, "Reactor Vessel Level Indicating System," on the installed RVLIS system,

2) verification of the SMM installation,

3) verification that the CETs were environmentally qualified, and

4) verification of operator training.

The inspector reviewed Temporary Plant Test (TPT) 58 on the system. The installed RVLIS system was determined to be acceptable as documented in memo MHS-007-85, dated September 6, 1985. The inspector reviewed the test and the RVLIS system appeared to have been satisfactorily tested. The RVLIS system was also verified to be operating in the control room.

As noted above, RG 1.97 provided requirements for Post-Accident instrumentation including SMMs, CETs, and RVLIS. This instrumentation was reviewed against RG 1.97 in NRC inspection report 50-344/88-36. The previous inspection verified the adequacy of the instrumentation installation, including the SMM installation, and that the CETs were environmentally qualified.

The inspector verified that operators were knowledgeable on the use of the CETs, and that there were training manuals that described the SMM and RVLIS. This item is closed.

No violations or deviations were identified.

9. Surveillance Procedures and Program (56700, 61700)

The inspector audited the licensee's surveillance program by evaluating the effectiveness of surveillance scheduling, verifying that a selected number of T.S. surveillances were being conducted in accordance with approved procedures, and reviewing the surveillance procedure against the intent of the surveillance requirement. Additionally, the inspector evaluated the completeness of the documentation and the test results against the acceptance criteria.

The licensee schedules surveillances using a computerized program, Surveillance Monitoring System (SMS). Administrative Order (AO) 6-5, "Control and Use of the Surveillance Monitoring System (SMS)," defines the method to control and use the Surveillance Monitoring System. The licensee uses the SMS to ensure surveillances are performed when required. Computer memory maintains the frequency of performance and the last several times that the surveillances were performed. Surveillances' performance date may be extended up to 25% of the periodicity (1.25 criteria), but the last three surveillances cannot exceed 25% of one frequency interval (3.25 criteria), the same as mentioned in surveillance requirement 4.0.2. The SMS does not monitor conditional technical specification surveillance requirements nor include any Radiation Protection or Chemistry technical specification. The Radiation Protection and Chemistry groups manually manage the scheduling of surveillances under their cognizance.

The the following technical specifications (TSs) were evaluated:

Technical Specification	Title
3/4.1.1.1	Shutdown Margin

3/4.1.1.5	Minimum Condition for Criticality
3/4.4.4	Pressurizer Heaters and Water Level
3/4.4.7	Reactor Coolant System Chemistry
3/4.5.1	ECCS Accumulators
3/4.5.5	Refueling Water Storage Tank
3/4.6.2.2	Spray Additive System
3/4.6.4.4.	Hydrogen Mixing Systems
3/4.8.2.3	DC Electrical Distribution, Operating
3/4.8.2.4	DC Electrical Distribution, Shutdown

The following paragraphs document the inspector's findings.

a. TS 3.1.1.1, Shutdown Margin (SDM)

TS 3.1.1.1 ensures subcritically can be achieved for all operating conditions, and Surveillance requirement 4.1.1.1.1.a establishes the requirements for shutdown margin monitoring in the event of a stuck control rod. If the stuck control rod is determined to be untrippable, the shutdown margin is required to be determined every 12 hours in Modes 1 through 5. While history indicates no stuck rod occurances, the licensee's procedure for a stuck rod, Off-Normal Instruction (ONI) 2-4, "Control Rod and Rod Position Indication," does not require 12 hour surveillance monitoring for shutdown margin nor do any procedural steps to implement this conditional surveillance requirement exist for Modes 3, 4 or 5. The inspector was informed that the licensee plans by October 31, 1990, to modify ONI 2-4 to include the requirement to verify shutdown margin every 12 hours.

The inspector also evaluated the licensee's daily verification of the shutdown margin in Modes 3, 4 and 5. Specification 4.1.1.1.e has shutdown margin verified in Modes 3, 4, and 5 by consideration of six factors: 1) RCS boron Concentration, 2) Control Rod Position, 3) RCS Average Temperature, 4) Fuel Burnup, 5) Xenon Concentration, and 6) Samarium Concentration. Shutdown margin must be greater than 1.6% delta K/K. In PGE's calculation of SDM, three of the factors (Xenon concentration, Samarium concentration and RCS average temperature) are not explicitly calculated. Instead, licensed operators verify SDM by comparing the actual boron concentration to curves that collectively incorporate the above three factors. These curves are derived from the vendor reload analysis, which the reactor engineers verify during reactor physics tests that are performed after each refueling. Xenon and Samarium Concentrations were not considered separately since their effects are small compared to temperature effects. By not considering their effects, additional conservatism is added to the calculation. The licensee has two curves that refer to temperature, one titled Cold Shutdown and the other Hot Shutdown. The Cold Shutdown curve has 68 degrees F. as the reference RCS temperature and the Hot Shutdown curve uses 557 degrees F. (Hot Zero Power) as the RCS reference temperature. No intermediate temperature curves exists.

In Modes 3 through 5, plant operators verify the shutdown margin daily (every 24 hours) per POT 24-2, "Daily Operating Routine." POT

24-2 has the operator refer to Operating Instruction (OI) 11-8, "Shutdown Margin," which instructs the operator to determine the shutdown margin in Modes 3 and 4 using the Hot Shutdown curve as a basis. Modes 3 and 4 can be between 200 and 557 degrees F. Because this procedural guidance is not definitive on which curves to use and could result in determining the shutdown margin nonconservatively, SDM determination is a followup item (50-344/90-29-02).

b. 3/4.4.4 Pressurizer Heaters and Water Level

This T.S. 3/4.4.4 ensures that sufficient pressurizer heater capacity is available from emergency buses to maintain RCS pressure during a natural circulation cooldown. The TS also verifies pressurizer water level is within limits.

Pressurizer level is determined to be within its limits once per 12 hours by POT 24-1. Heater capacity is determined adequate once each refueling by performing POT 1-4, "Pressurizer Heater Functional Test." The inspectors review of this test identified the POT did not contain the appropriate criteria to verify required heater capacity (150 kw). The acceptance criteria in POT 1-4 measured only heater current and stated that "a total of 185 amps corresponds to 150 kw at 460 volts." The product of the square root of three times the voltage and current calculates the power generated of a three phase alternating current resistance heater. Consequently, the acceptance criteria stated in POT 1-4 assured the availability of 147.4 kw vice 150 kw.

In reviewing the data sheets for POT 1-4, the inspector determined available heater capacity had always been greater than 150 kw because bus voltage was usually 480 $\pm$ 5 volts instead of the 460 volts assumed in the surveillance and the amperage was substantially in excess of 185 amps.

The inspector discussed this with the licensee and the licensee wrote Corrective Action Request (CAR) 90-5326. In CAR 90-5326, the licensee determined that the present acceptance criteria was inserted during the last procedure upgrade. In an attempt to quantify the acceptance criteria, the procedure writer made a math error. This change was not part of the licensee's review process and was not noticed. The licensee implemented Temporary Change Notice (TCN) 90-136 to raise this acceptance criteria to 225 amps.

10 CFR Part 50, Appendix B, Criterion V states, in part, that "Instruction, procedures, or drawings shall include appropriate quantitative or qualitative acceptance criteria for determining that important activities have been satisfactorily accomplished." The acceptance criteria in the procedure was not appropriate. Due to the licensee's prompt corrective actions with respect to this item, and the minor safety significance, the violation is not being cited because the criteria cited in Section V.A of the enforcement policy were satisfied.

c. <u>3/4.6.4.4 Hydrogen Mixing Systems</u>

The hydrogen mixing system was provided to ensure adequate mixing of the containment atmosphere following a Loss of Coolant Accident (LOCA). This mixing will prevent localized accumulations of hydrogen. The surveillance requirements are, quarterly on a staggered test basis, start the system and verify it operates, and, once per 18 months, verify a system flow rate of 2500 $\pm$ 250 cubic feet per minute (cfm).

The licensee's verification of the flow rate is performed as part of Periodic Engineering Test (PET) 10-1, "Air Purification and Cleanup Systems." This procedure was last revised on March 29, 1990, and section 8.3 of the test verifies system flow.

PET 10-1 had an incorrect technical specification reference (Table I) pertaining to the Hydrogen mixing system. The procedure referred to the wrong data sheet for recording the flow rate. The test was performed on March 29, 1990, and this error was not discovered during that testing. However, the system was verified to have met its design flow rate. The licensee is in the process of correcting the procedure.

d. <u>3/4.8.2.3 DC Electrical Distribution, Operating</u>

Technical specification 3/4.8.2.3 verifies the operability of the station batteries. The station batteries are surveillance tested weekly, quarterly, each refueling and once very five years. To verify the direct current (DC) bus operability, the licensee uses Maintenance Procedure (MP) 1-14, "125 Volt Station Batteries," and MP 1-15, "130-Volt Station Battery Chargers." In reviewing the procedure against the technical specification, the inspector identified the following discrepancies:

Technical specification references are not precise for MP 1-15. This procedure states that it meets TS 4.3.2.3.2. In fact, it verifies only part of the specification, TS 4.3.2.3.2.c.3.

Both procedures state that "all sections preceded by an asterisk (*) are Technical Specifications required surveillance items, and all Technical Specification acceptance criteria are identified by a double asterisk (**) either in the procedure or on the required data sheet." In fact, though the procedures do cover most of the Technical Specification requirements, not all of the Technical Specifications have an asterisk in the procedure. Specifically, the 18 month surveillance requirements were not asterisked in MP 1-14.

Technical Specification 4.8.2.3.2.c.2 states that once every 18 months inspections are conducted to verify that "The cell-to-cell and terminal connections are clean, tight, free of corrosion and coated with anti-corrosion material." The Technical Specification for verifying the cell connections tightness was not implemented in the procedure. The licensee does verify the intercell resistances and compares them to the original installation but does not physically verify tightness. At the time of the inspection, the licensee was in the process of upgrading MP 1-14. The new procedure does include steps to verify cell-to-cell tightness to implement the technical specification.

The battery service test profile in MP 1-14 is different from the FSAR battery service test profile. The service test profile is described in Calculations TE-119, "125-V Batteries," and TE 120, "125-V Service Test." The profile in TE 120 is the station profile used in the test. The profile stated in the FSAR also is not the same as in the calculation. It appears the FSAR was not adequately updated following the changeout of station batteries in 1988. The licensee is evaluating the Request for Design Change (RDC) associated with the battery replacement.

In summary, this Surveillance Program audit identified weaknesses with the technical adequacy of selected surveillance procedures and the implementation of some conditional technical specifications.

One Non Cited Violation (NCV) was identified and one followup item identified.

10. Exit Interview (30703)

The inspectors met with the licensee representatives denoted in paragraph 1 on October 15, 1990 and with licensee management throughout the inspection period. In these meetings the inspectors summarized the scope and findings of the inspection activities.